12th International Conference on Emerging Nuclear Energy Systems (ICENES'2005)
Brussels, Belgium, August 21–26, 2005, on CD-ROM, SCK•CEN, Mol, Belgium (2005)

SUPERCRITICAL LIGHT WATER REACTOR (SCLWR) WITH INTERMEDIATE HEAT EXCHANGER (IHX)

Prof. W. Van Hove
Tractebel Engineering, division of Suez
Arianelaan 7, B-1200 Brussel
walter.vanhove@tractebel.com

Prof. R. Beauwens and student J. Gorgemans
Service de Métrologie Nucléaire
Faculté des Sciences Appliquées
Université Libre de Bruxelles (ULB)
ULB CP165/84, avenue F.D. Roosevelt 50, 1050 Bruxelles
robert.beauwens@ulb.ac.be ; julie.gorgemans@gmail.com

Prof. M. Giot and student G. Donnet
Unité de Thermodynamique et Turbomachines
Faculté des Sciences Appliquées
Université Catholique de Louvain (UCL)
Place du levant 2, B-1348 Louvain-la-Neuve
giot@term.ucl.ac.be ; gdonnet@gmail.com

ABSTRACT

In this paper, a SCLWR concept with an IHX is proposed. Like the SG in a PWR, the IHX separates the primary loop from the secondary loop. The primary loop can then be completely enclosed within the reactor building. Such a concept will inevitably lead to a higher investment. But the advantage is that all primary activity remains within the primary loop and within the reactor building and no contamination of the turbine and BOP occurs. Moreover, this concept allows a separate chemistry for the primary and secondary loops. It also allows the use of a soluble neutron poison for reactivity control.

A conceptual design of the reactor vessel and the IHX is proposed; a RELAP model of the primary and secondary systems has been built and some design base accidents have been analysed using the RELAP5/mod3.3 code. These analyses are performed to investigate the general behavior of a SCLWR with IHX, to have an idea of the grace time available before fuel damage occurs and to obtain some indication such as which type of safety systems would be needed for this concept. The purpose of the exercise is to determine whether the advantages (if any) of this concept with IHX sufficiently outweigh its drawbacks and consequently whether it is worth pursuing the development of this concept with IHX.

Key Words: light water reactor, supercritical pressure, generation IV, RELAP, safety analysis

1 INTRODUCTION

The very first developments of a supercritical light water reactor (SCLWR) date back to the late 1950's, early 1960's, see historical overview in [1,2]. In the late 1960's, there were also some attempts to design a steam cooled fast reactor. But the rapid and spectacular development of the LWR's (both PWR and BWR) on the one hand and the important R&D that was required to develop such a SCLWR on the other hand, stopped their further development and the idea of a SCLWR was abandoned for quite some time. But in fossil-fired power plants the development continued and supercritical boilers have been in operation for over 30 years now.

In the late 1980's, the Kurchatov institute took up the idea again and proposed a concept of a small, integral type PWR with the primary loop operating at supercritical conditions [3]. But the concept of a SCLWR was really revived by the University of Tokyo in the 1990's [4-9]. The important novelty in the concept of the University of Tokyo was that they used the BWR as starting point for their development. This led to important simplifications and cost savings with respect to current PWR's, while at the same time overall cycle efficiency was increased.

Following the work at the University of Tokyo, several R&D projects were launched:

1. In Canada, the CANDU-X project by AECL to study supercritical versions of the CANDU reactor;
2. In the US, some smaller projects were financed within NERI by the US DOE;
3. In Europe, the HPLWR project [10] was funded by the EC 5th framework program.

In 2002, the Generation IV project selected the SCLWR as one of the six most promising concepts for future nuclear reactors. Since then, there is a worldwide renewed interest in SCLWR's and a large, international R&D program is launched within GenIV.

The work at the University of Tokyo clearly demonstrated the main advantages of the SCLWR concept with respect to current LWR's:

1. Increased thermal efficiency, leading to reduced fuel cost and waste disposal cost per kWh produced;
2. Important plant simplifications and consequently a reduced investment cost per installed kWe;
3. Possibility of both a thermal and a rapid neutron spectrum core.

The main viability issue of the SCLWR concept is the selection of suitable materials. The most difficult problem is probably the fuel rod cladding material. Besides the material problem, core design is also a major viability issue. Due to the large enthalpy rise over the core, the cladding temperature is very sensitive to the hot channel factors, making core design a very difficult task.

All recently proposed SCLWR concepts are once-through or direct-cycle designs, based on the BWR concept, where the supercritical water from the reactor is directly fed to the turbine. This concept excels in system simplification and cycle efficiency, but the drawback is that the turbine and the BOP inevitably become activated. In a BWR, soluble and suspended radioactive products remain in the reactor by the process of separation between water and steam. In a SCLWR however, all activity in the supercritical fluid is transported to the turbine. A leaking fuel rod will in this situation inevitably lead to an immediate reactor shutdown. Current PWR's on the other hand can continue to operate with several leaking fuel rods till the next planned outage.

In this paper, a SCLWR concept with an Intermediate Heat Exchanger (IHX) is proposed. Like the SG's in current PWR's, the IHX separates the primary loop from the secondary loop or power conversion system. The primary loop can than be completely enclosed within the reactor building. Such a concept will lead to a higher investment cost and a higher reactor temperature for the same cycle efficiency. But the advantage is that the whole primary activity remains within the primary loop and within the reactor building and no contamination of the turbine and BOP occurs. This concept also allows a separate chemistry for the primary and secondary loops and allows the use of a soluble neutron poison for reactivity control. Moreover, it is hoped that this concept will behave much like a current PWR during design base accidents, so that the vast experience in this field with today's PWR's can to a large extend be recuperated.

A conceptual design of the reactor vessel and the IHX is proposed, a RELAP model of the primary loop has been built and some design base accidents have been analyzed using the RELAP5/mod3.3 code. These analyses are performed to investigate the general behavior of a SCLWR with IHX, to have an idea of the grace time available before fuel damage occurs and to obtain some indication such as the type and capacity of the safety systems that would be needed for this concept.

The purpose of the whole exercise is to determine whether the advantages (if any) of this concept with IHX sufficiently outweigh its drawbacks and consequently whether it is worth pursuing the development of this concept with IHX.

2 CONCEPTUAL DESIGN

2.1 Operating Conditions

The operating conditions are given in Table I. These operating conditions are those of the conceptual design for a direct-cycle SCLWR developed by INEEL [11]. The reference design of the power conversion cycle in [11] considers turbine inlet conditions of 25 MPa, 500 °C and generates 1600 MWe with a net thermal efficiency of 44,8 %. The corresponding core thermal power is 3575 MWt.

In order to preserve the performance of the power conversion cycle, the same operating conditions were kept on the shell side of the IHX. The reactor operating temperatures were consequently increased with 20 °C. The reactor operating pressure was also increased to 28 MPa. This has a favorable impact on the size of the IHX, see section 2.4.

Table I. Operating Conditions

Primary system		**Power conversion system**	
Core power	3575 MWt	**Net electric power**	1600 MWe
Reactor inlet temperature	300 °C	**Turbine inlet temperature**	500 °C
Reactor outlet temperature	520 °C	**Turbine inlet pressure**	25 MPa
Reactor operating pressure	28 MPa	**Feedwater temperature**	280 °C
Reactor flow rate	1916 kg/s	**Feedwater flow**	1847 kg/s

2.2 Core and Fuel Assembly Design

Recent papers on SCLWR core design seem to converge to a square lattice Fuel Assembly (FA) design with water rods [8,9,12,13,14] for a thermal spectrum reactor. The core and FA design proposed in [11] is adopted. The core design parameters are given in

Table II. The FA and fuel pin relevant dimensions are given in Table III and Table IV. The reference core is shown in Fig. 1, the FA cross section in Fig. 2.

Table II. Reference Core Design, from [11]

Number of fuel assemblies	145
Fuel assembly type	square 25x25 array
Fuel assembly pitch	0,288 m
Core Barrel inner/outer diameter	4,3/4,4 m
Axial/Radial/Local/Total Power Peaking Factor	1,4/1,4/1,2/2,35
Average/Peak Linear Power	192,6/453,0 W/cm

Table III. Reference Fuel Assembly Design, from [11]

Assembly side	286 mm
Assembly duct thickness/material	3 mm/MA956
Number of fuel pins/water rods	300/36
Fuel pin pitch	11,2 mm
Water rod side	33,6 mm
Water rod thickness/material	1 mm/MA956 + 2 mm/Zirconia

Table IV. Reference Fuel Pin Design, from [11]

Fuel pin outside diameter	10,2 mm
Cladding thickness/material	0,63 mm/MA956
Fuel pellet outside diameter	8,78 mm
Fuel column length	14 ft / 4,2672 m
Gas plenum length	0,6 m
Fill gas pressure at ambient conditions	6,0 MPa

Part of the coolant flows downwards through the water rods to provide sufficient moderation in the core. In the bottom nozzle of the FA, this flow is mixed with the remaining fraction and the total amount of coolant flows upwards to cool the fuel pins. In the literature widely different values are proposed for the downflow fraction. In this paper, the value of 90 % proposed in [11] is adopted. Because the heating of the downward flowing water in the water rods has an important impact on its density, the value of the downflow fraction can only be optimized using coupled neutronic and thermal-hydraulic calculations.

For the cladding material, [11] suggests the use of the oxide-dispersion strengthened ferritic alloy Incoloy MA956. This is a Fe-Cr-Al steel, mechanically alloyed with Yttrium oxide particles. This alloy has excellent oxidation resistance and high creep strength up to temperatures as high as 1300 °C. Compared to Ni based alloys such as Inconel MA754, which shows similar strength at elevated temperatures, MA956 has a small advantage in neutron economy because it is Ni free. No data are available on the behavior of MA956 under irradiation. MA956 has been assumed as cladding material and as structural material for the fuel assemblies in this concept.

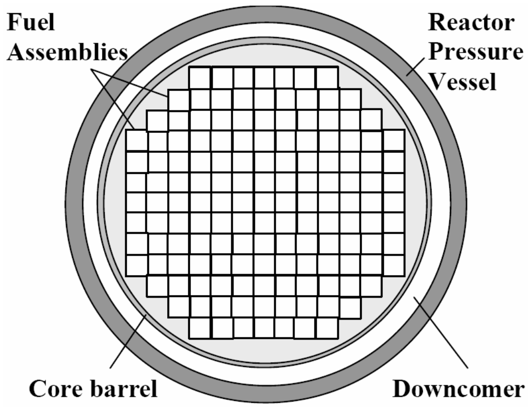

Figure 1. Core Layout, from [11]

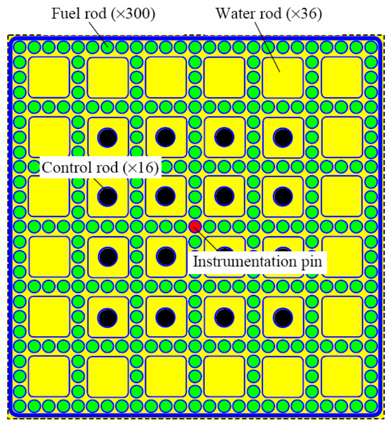

Figure 2. Fuel Assembly Design, from [11]

To avoid unacceptable cladding temperatures during normal operation, the coolant flow through each FA must be proportional to the FA power. This can be achieved by placing orifices at the outlet of each FA to obtain a flow distribution that matches the power distribution. This also requires that the fuel assemblies are ducted. No cross flow between FA can be allowed.

The original FA design in [11] proposed a water rod wall thickness of 0,4 mm, all metal. But in [11], the need for insulated walls was already suggested. Test calculations with parallel fuel assemblies of different powers indicated that it was not possible to obtain a stable flow distribution proportional to the FA power with the thin, all metal wall for the water rods. The heat transfer through this thin wall is very high and in the higher powered fuel assemblies, the water inside the water rods can be heated to temperatures above the pseudo-critical temperature while flowing downwards. This leads to large density changes inside these water rods resulting in an unstable flow behavior and flow reversal. To obtain a stable flow distribution proportional to the FA power, it is necessary to keep the temperature inside the water rods well below the pseudo-critical temperature. This requires insulation of the water rod walls.

A suitable insulating material might be stabilized Zirconium oxide or Zirconia. This ceramic material combines a high strength at elevated temperatures with low heat conductivity. The use of Zirconium also has the advantage of low neutron absorption. Therefore, an insulating insert of 2 mm Zirconia is assumed inside the water rods. Also the prolongation of the water rods through the upper plenum of the reactor vessel is assumed insulated with 2 mm of Zirconia. A temperature difference of over 200 °C develops over this thin Zirconia wall during normal operation. The resulting thermal stresses might be a problem for this thin wall. Some information on the behavior of Zirconia under irradiation is available from tests with uranium free nuclear fuel. Little is known on the long term behavior of Zirconia in an aqueous environment.

2.3 Reactor Vessel Design

Compared to a typical large-size PWR vessel, the SCLWR vessel must accommodate the additional requirement that a significant part of the cold leg flow must be directed to the volume below the reactor vessel head and then flows downwards to feed the water rods. In a typical PWR vessel layout, this would require a very complex construction at the level of the FA top nozzle and upper core plate, where the colder fluid coming from the dome and feeding the water rods must be separated from the hot fluid coming out of the FA and flowing counter-currently towards the upper plenum and the hot legs.

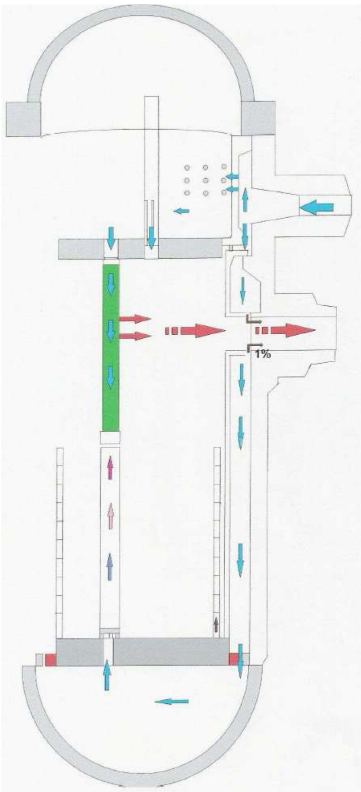

Figure 3. Reactor Vessel

To avoid this difficulty, an alternative reactor vessel design is proposed in Fig. 3. The hot and cold legs are connected to the reactor vessel at different elevations, the cold leg having the higher elevation. The upper core plate has been eliminated altogether. Instead, the FA duct is prolonged with 1,8 m, bringing the total FA length to about 7,1 m. The FA is positioned between the lower and upper core support plates. The sides of the duct prolongation are perforated so that the hot fluid from the FA can escape sideways towards the hot legs. Each water rod in Fig. 2 is prolonged with a circular tube (with 2 mm Zirconia insert) running inside the duct prolongation up to the upper core support plate.

Most of the water coming from the cold legs flows immediately to the volume under the dome through the perforated skirt of the upper internals. The upper core support plate has an opening corresponding with each FA. The water flows through these openings into the water rods and flows downwards to the FA bottom nozzle. The remaining part of the coolant flows downwards between the barrel and the vessel to the lower plenum. The lower core support plate also has an opening for each FA. Inside the FA bottom nozzle, the two fractions are mixed. The coolant then flows upwards along the fuel pins and escapes sideways out of the duct prolongations towards the hot legs.

Just as in a typical PWR vessel, a control rod guide tube is placed on top of a selected number of FA openings in the upper core support plate. A penetration in the vessel head is aligned with each control guide tube for the control rod drive shaft. The control rod clusters are largely the same as in a typical PWR vessel, only the rods are now considerably longer because of the duct prolongation. Note that the control rods are guided inside the water rods and are therefore not in contact with the high temperature water.

The flow path through the reactor is such that the pressure retaining boundary (vessel + head) remains at the cold leg temperature of 300 °C. This allows the use of current state-of-the-art LWR vessel materials with a stainless steel cladding on the inside. The thickness will however be significantly larger because of the much higher design pressure compared to a typical PWR vessel. For those internals exposed only to the cold leg temperature, the same austenitic stainless steels used for the internals of current PWR vessels can be used. For the internals exposed to higher temperatures, the ferritic-martensitic 9Cr steels like the P91 or P92 could be used. These materials are extensively used in supercritical fossil-fired power plants for the high temperature components. But no data are available on the evolution of the mechanical properties of these materials under irradiation. In this model SA 508 Grade 3 Class 1 steel has been assumed as vessel material and the P91 9Cr-1Mo-V steel has been assumed for the internals.

2.4 Intermediate Heat Exchanger Design

The IHX design is inspired by the once-through Steam Generator design of B&W, as illustrated in Fig. 4. The primary fluid is on the tube side, the secondary fluid on the shell side.

Table V. IHX Main Dimensions

Tubes OD/wall thickness	12,7/1,7 mm
Number of tubes	15475
Tube bundle length	24,78 m
Inner/outer radius of tube bundle	0,2/2,142 m
Helical angle	20°
Total heat transfer area	44730 m²

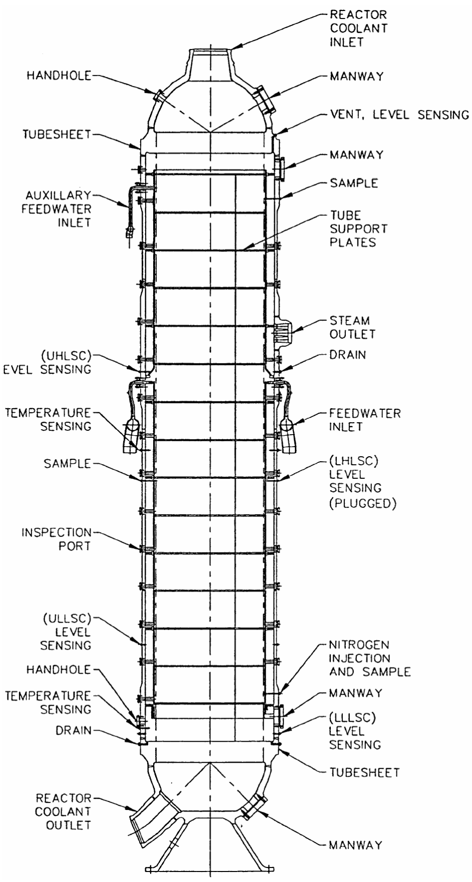

Figure 4. Once-through SG, from [15]

Sizing calculations have been performed for three types of IHX: straight tubes pure counter-current (like in Fig. 4); baffled design with cross flow on the shell side; helically wound tube bundle. The helically wound tube bundle clearly gives the best performance for this application. According to these calculations, it seems possible to transfer the heat using only two helically wound IHX's with dimensions of the same order of magnitude as a typical once-through SG. Incoloy 800HT was assumed as tube material. The main dimensions of the IHX as used in this concept are given in Table V.

The sizing calculations were performed using the Dittus-Boelter heat transfer correlation [2,16] on the tube side and the Zukauskas correlation [17] on the shell side. It is acknowledged that the Dittus-Boelter correlation is not the most appropriate correlation and overestimates the heat transfer coefficient around the pseudo-critical point [2,16]. As for the Zukauskas correlation, no information is available on its applicability for supercritical fluids. It uses however a form similar to the Dittus-Boelter correlation and it can therefore be assumed that it suffers from the same shortcomings. To the authors' knowledge however, no other correlation is available for the heat transfer on a helical tube bundle in supercritical fluids. The two correlations mentioned were selected for the sizing calculations because they require knowledge of the fluid properties only at the bulk fluid temperature and allow therefore a direct calculation without iteration on the wall temperature. This greatly simplifies the sizing calculation. Notwithstanding the known shortcomings, it is believed that a reasonable first guess of the size of the IHX is obtained, which is judged to be sufficient for the purpose of this paper.

Because a helical tube bundle is used, the differential thermal expansion between tubes and shell poses no problem. The feedwater inlet is therefore moved to the top of the IHX shell and the feedwater flows downwards between bundle wrapper and shell. In this way the outer shell is only exposed to the feedwater temperature, which allows a smaller shell thickness. Nevertheless, wall thickness will be much larger than in Fig. 4 due to the much higher design pressure.

The calculated temperature evolution on the shell and tube side is given in Fig. 5. This figure highlights a specific problem of heat exchangers with supercritical fluid where the fluid crosses the pseudo-critical temperature during heating or cooling. It is observed that around the pseudo-critical temperature, a zone of nearly constant temperature develops. This is due to the very high specific heat of the fluid around this temperature. In this region, nearly all of the transferred heat is used to overcome the high thermal capacity and the resulting temperature change is very small. This behavior is somewhat similar to a boiling system. Therefore the pseudo-critical temperature must be sufficiently higher on the primary side than on the secondary side. If not, the temperature difference becomes very small in the pseudo-critical region, resulting in very low heat transfer and therefore requiring a much larger heat transfer area. For the same reason, it is not possible to operate the primary system at a lower pressure than the secondary system.

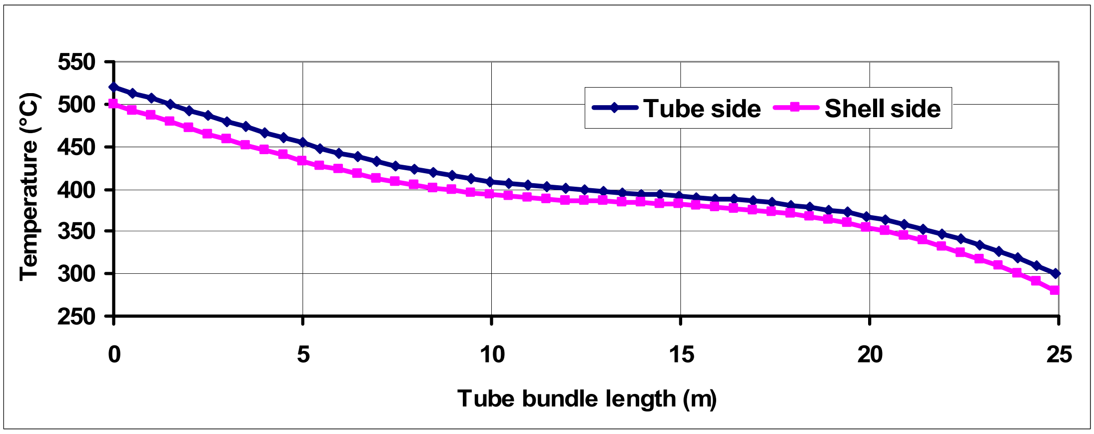

Figure 5. Calculated Temperature evolution in IHX

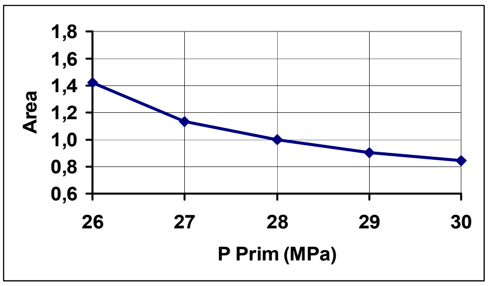

Figure 6. Relative HX Area vs. Primary Pressure

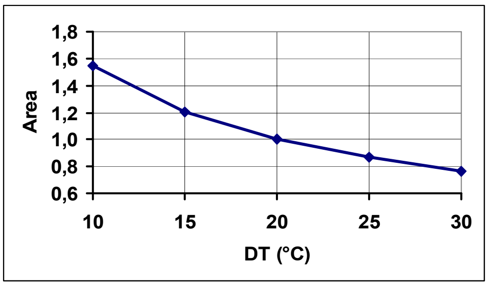

Figure 7. Relative HX Area vs. ΔT

The impact of the primary pressure on the required heat transfer area is shown in Fig. 6. The impact of the primary to secondary temperature difference on the heat transfer area is likewise shown in Fig. 7. The chosen primary system operating conditions are a reasonable compromise, but there is certainly room for optimizing the IHX dimensions.

2.5 Primary Loops and Coolant Pumps

The layout of the primary loops is inspired by the AP1000, which also uses two SG's. But with this difference of course that the hot leg runs to the top of the IHX, like in the B&W plants. Both layouts are illustrated in Fig. 8 and 9.

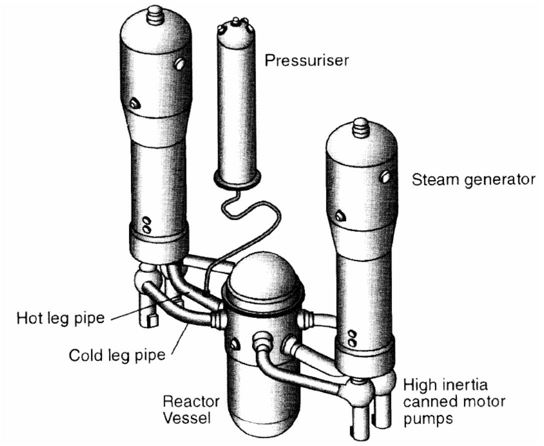

Figure 8. AP1000 Primary Loops, from [18]

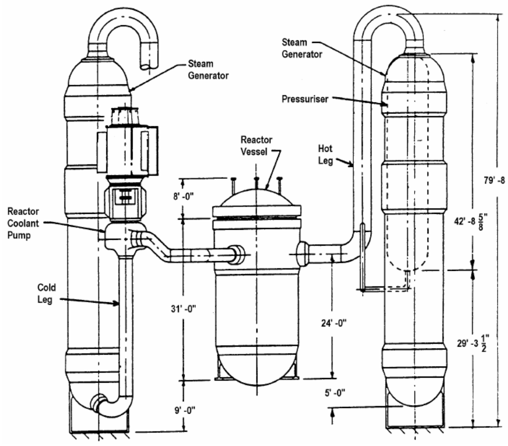

Figure 9. B&W Primary Loops, from [19]

There are 4 cold legs and 2 hot legs. The 9Cr steel T91 has been assumed for the pipes.

High inertia canned motor pumps are assumed, 2 pumps per IHX. The pumps are roughly half the size of the AP1000 pumps. Design temperature is about the same as in the AP1000 design, but the design pressure would be much higher. The inertia is assumed to be about ¼ of the AP1000 pump inertia.

3 RELAP MODEL

A RELAP model has been built of the reactor vessel, both primary loops, both IHX's and the feedwater and steam lines up to the isolation valves. Accumulators, an auxiliary feedwater system and a limited number of reactor shutdown signals were also added to the model. The code version used is RELAP5/mod3.3ey.

The adequacy of the RELAP5/mod3 code to calculate transients in the supercritical regime has been investigated in [20]. The authors of [20] conclude that, although the heat transfer correlations are not the most appropriate for supercritical conditions, the overall prediction capability of RELAP5/mod3 is sufficient to investigate the general behavior of a system with supercritical water as long as the pressure remains supercritical. But the code version used in [20] failed whenever the pressure had to cross the critical pressure during depressurization transients. The code version used in this paper still suffers from the same problem. For this reason, no Loss Of Coolant Accident (LOCA) or Steam Line Break (SLB) scenarios could be calculated so far.

During the course of this project, the authors were in close contact with the code developers at ISL Inc., reporting any code problem that was encountered. This resulted in a number of modifications to the 3.3ey version to improve the code performance in the supercritical regime. However the problem of crossing the critical pressure is not yet resolved and the code developers are currently working on it.

Many quantities vary strongly along the core height e.g. coolant temperature and density, fuel and cladding temperatures. These variations are also highly non-linear. To capture these variations correctly, a fine mesh in the axial direction of the core is needed. In this model, 21 nodes along the heated length of the fuel were used. The hottest FA including the hottest fuel rod has also been modeled explicitly in parallel with the averaged core. A chopped cosine power distribution with the power peaking factors as in table II is used.

A special problem occurred when simulating heat exchangers with small temperature differences as in Fig. 5. Like all nodal codes, the RELAP code basically uses the energy balance over the node to calculate the temperature in the node. But this means that the averaged node temperature is in reality the node outlet temperature. In a counter-current heat exchanger, this results in an even smaller temperature difference between tube and shell side than in reality. The only possibility to obtain a reasonable simulation of the heat exchanger is to use a very fine meshing along the tube bundle. In this model, 60 nodes on either side of the tubes were used to represent the tube bundle.

The RELAP model of the IHX was first calibrated to reproduce the calculated design conditions. The vessel model was also calibrated to obtain the correct flow distribution between averaged and hot FA, the necessity of which was discussed in section 2.2. Next, the model was assembled and the pumps calibrated to give the correct primary loop flow. A satisfactory steady-state solution at nominal power was obtained for the complete model as starting point for the accident analysis. This steady-state solution reproduces the operating conditions of Table I. Resulting hot rod maximum cladding temperature is 625 °C and occurs at the top of the fuel rod. Maximum fuel centerline temperature is 2540 °C in the middle of the hot rod. Required pump head is 0,74 MPa, pump power is 625 kW.

4 DESIGN BASE ACCIDENTS

4.1 Loss of one Reactor Coolant Pump

The first accident analyzed is the loss of one out of four reactor coolant pumps. The results are shown in Fig. 10.

Pump trip is postulated at t = 5 sec in the transient. Reactor shutdown occurs on low pump speed, set at 90 % of nominal speed. The core power reduction is simulated using a power curve in function of time following reactor shutdown, taken from a typical PWR with comparable core power. The reactor shutdown also causes turbine trip and loss of feedwater. The primary system relief valves open at 30,0 MPa, and the steam line relief valves open at 26,5 MPa. The primary system relief valves are assumed connected to the top elevation of both hot legs.

Following the pump trip, the mass flow in the corresponding cold leg rapidly decreases and becomes negative about 8 sec after pump trip. The mass flow in the other cold leg of the same loop increases with about 40 %. The mass flow in the other loop increases with about 14 %. After reactor shutdown, the heat evacuated by the IHX's rapidly becomes larger than the core power, assuring the cooling of the primary loop. The pressure in the primary system rises and the pressure relief valves must open during about 15 sec to limit the primary pressure. A total of about 500 kg of primary fluid is discharged. The secondary pressures are controlled by the steam relief valves.

The increase of the cladding temperature is rather limited. The maximum cladding temperature of the hot rod increases with only 35 °C and reaches a maximum of 660 °C.

Clearly the accidental loss of one reactor coolant pump poses no problem for the cladding temperature, provided that reactor trip occurs on low pump speed or low primary flow.

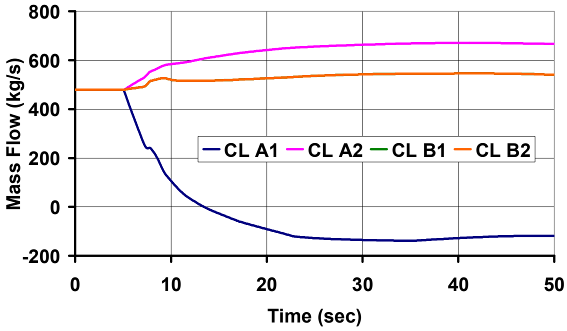

Figure 10a. Cold Leg Flow

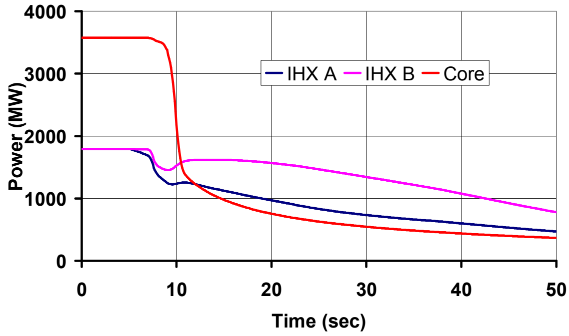

Figure 10b. Core and IHX Power

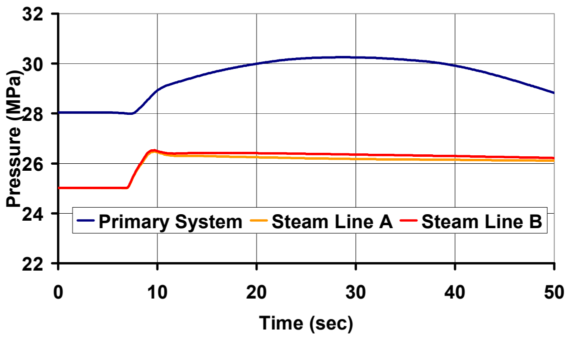

Figure 10c. Primary and Secondary Pressure

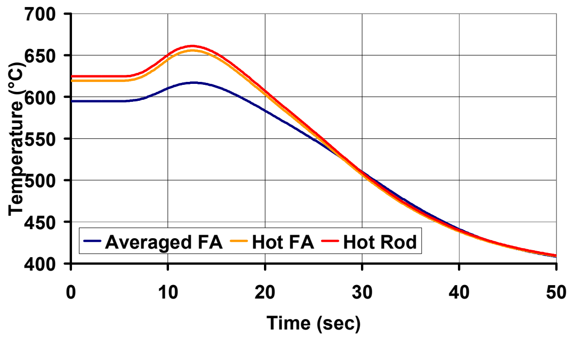

Figure 10d. Maximum Cladding Temperature

4.2 Blackout

The second accident analyzed is the blackout or loss of non-emergency electrical power. This results in the simultaneous loss of all four reactor coolant pumps, turbine trip and loss of feedwater. The results are shown in Fig. 11 for the short term and Fig. 12 for the long term.

In the short term, it is verified that the reactor shutdown signals are capable of protecting the core against too high cladding temperatures. This accident also dimensions the capacity of the pressure relief valves to avoid overpressure in the primary system. In the long term, it must be checked that the core can be adequately cooled by natural circulation.

The blackout is postulated at t = 5 sec in the transient and results in turbine trip, loss of feedwater and coast down of all four reactor coolant pumps. The reactor shutdown is delayed until the signal on low pump speed, set at 90 % of nominal speed. In this way, the analysis also covers the loss of all reactor coolant pumps accident. The core power reduction is simulated using a power curve in function of time following reactor shutdown, taken from a typical PWR with comparable core power. The primary system relief valves open at 30,0 MPa, and the steam line relief valves open at 26,5 MPa.

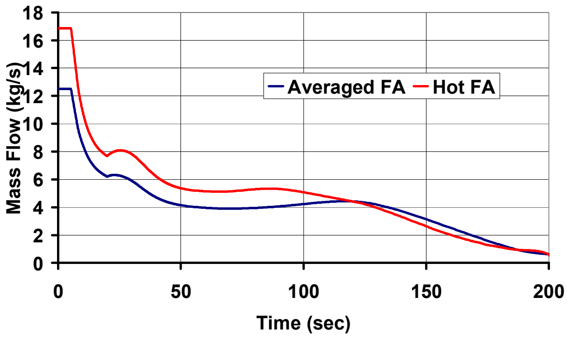

Figure 11a. Fuel Assembly Flow

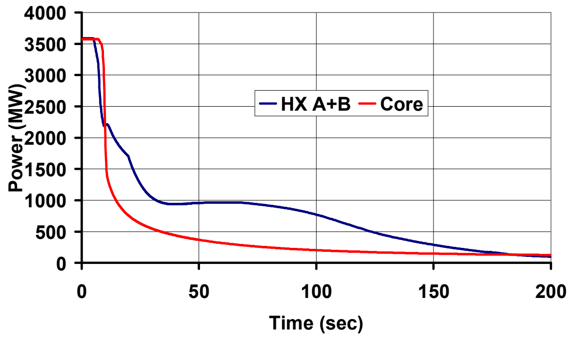

Figure 11b. Core and IHX Power

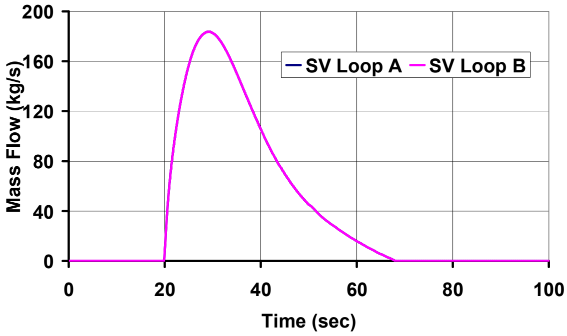

Figure 11c. Primary Safety Valve Discharge

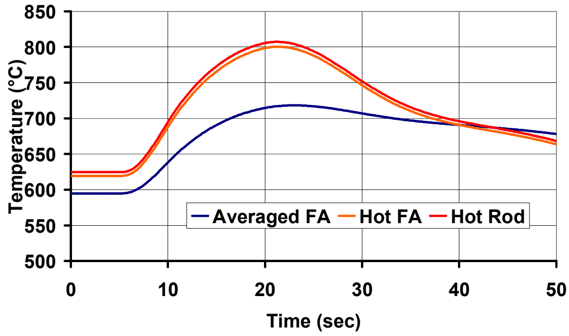

Figure 11d. Maximum Cladding Temperature

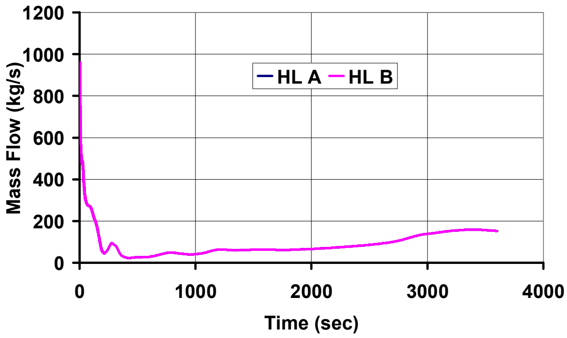

Figure 12a. Hot Leg Flow

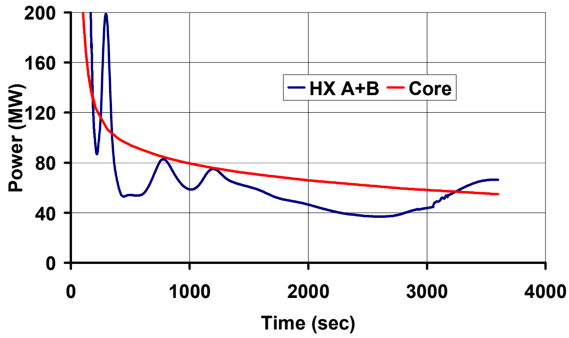

Figure 12b. Core and IHX Power

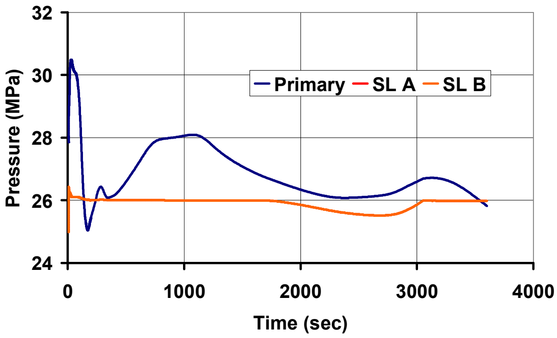

Figure 12c. Primary and Secondary Pressure

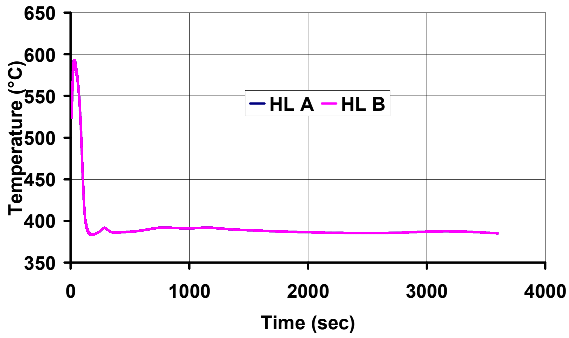

Figure 12d. Hot Leg Fluid Temperature

Following the pump trip, the mass flow through the FA's rapidly decreases, leading to a heat up of the system, a pressure increase and an increase of the cladding temperature. The core power is rapidly interrupted by the reactor shutdown signal on low pump speed or low primary flow. Maximum cladding temperature reached is 807 °C. At these temperatures, the ultimate tensile strength of the cladding material is still over 100 MPa (in unirradiated condition), which should be sufficient to withstand the loads under these conditions.

The primary system pressure relief valves open at 15 sec after the start of the transient and remain open for about 50 sec. Maximum relieving capacity is 370 kg/s and a total of nearly 8 t of fluid is discharged. The maximum pressure reached in the primary system is 30,5 MPa. Clearly, the SCLWR concept will require a much larger capacity for the pressure relief valves than a PWR of comparable core power.

Following turbine trip, the secondary side pressure rapidly increases until the steam line relief valves open. For the remainder of the transient, steam pressure is controlled by the steam relief valves and remains nearly constant.

After this initial phase, the temperature in the primary system rapidly decreases, but then remains nearly constant around the pseudo-critical temperature. A natural circulation flow develops in the loops. However the heat evacuated by the IHX's is at best about equal to the core power and no real cooling of the primary system is observed. Similar to a boiling system, the primary pseudo-critical temperature remains "stuck" slightly above the secondary pseudo-critical temperature.

For the cooling to really set in, it is necessary to bring cold AFW water into the IHX. An AFW flow of 25 kg/s to each IHX is assumed in this model. This AFW flow is injected in the feedwater line upstream of the IHX as is the usual practice in PWR's. But this means that about 1 h after the start of the transient, this cold AFW water has not yet reached the tube bundle. To improve the cooling of the primary system after a loss of feedwater, it will probably be necessary to inject the AFW directly in the bottom of the shell side. An optimization of the AFW flow and of the injection point will be necessary in order to obtain a correct cooling of the primary system after a loss of feedwater accident.

5 CONCLUSIONS

A concept of a SCLWR with IHX's has been presented. A reactor vessel concept has been proposed that allows a large fraction of the cold leg flow to feed the water rods in downflow without too much complication. Sizing calculations have been performed for the IHX, which indicate that it should be possible to transfer the core power with only two IHX. But further optimization of the dimensions of the IHX is necessary and perhaps a solution with three or four smaller heat exchangers might be more economical. A RELAP model of the entire system has been built and calculations have been performed with the RELAP5/mod3.3ey version of the code. Satisfactory simulations of the steady-state power operating conditions have been obtained. Unfortunately, the code version used fails when depressurizing the system below the critical pressure. Therefore, no LOCA or SLB transients could be studied so far and only the blackout scenario was investigated. Cladding temperatures remain acceptable in case of a loss of all circulation pumps. Required pressure relief valve capacity to avoid overpressure in the primary system will be higher than in a PWR of comparable core power. A natural circulation flow does develop in the primary loop in case of loss of all circulation pumps. However the long term cooling of the system with the IHX and the AFW is not straightforward and will require some careful optimization of the AFW system.

6 ACKNOWLEDGMENTS

Noting that the SCLWR is absolutely not within Tractebel Engineering’s core business, the authors wish to thank the director of TE’s Nuclear Department for allocating the budget and the resources that made it possible to present this challenging subject to the students as a Master Thesis. Special thanks also to Glen A. Mortensen from Information Systems Laboratories (ISL) Inc., USA, who made a great effort in resolving the code problems that were encountered all along this project.

7 REFERENCES

1. P. Dumaz et al., “Les réacteurs à eau à pression supercritique,” *Revue Générale Nucléaire*, **N° 4 - Juillet-Août**, pp.57-65 (2004).
2. I.L. Pioro, H.F. Khartabil, R.B. Duffey, “Heat transfer to supercritical fluids flowing in channels – empirical correlations (survey),” *Nuclear Engineering and Design*, **230**, pp.69-91 (2004).
3. V.A. Silin, V.A. Voznesensky, A.M. Afrov, “The Light Water Integral Reactor with natural circulation of the coolant at supercritical pressure B- SKDI,” *Nuclear Engineering and Design*, **144**, pp.327-336 (1993).
4. Y. Oka, S. Koshizuka, T. Yamasaki, “Direct Cycle Light Water Reactor Operating at Supercritical Pressure,” *Journal of Nuclear Science and Technology*, **29**, pp.585 (1992).
5. Y. Oka, S. Koshizuka, “Concept and Design of a Supercritical-Pressure Direct-Cycle Light Water Reactor,” *Nuclear Technology*, **103**, pp.295 (1993).
6. T. Nakatsuka, Y. Oka, S. Koshizuka, “Control of a Fast Reactor cooled by Supercritical Light Water,” *Nuclear Technology*, **121**, pp.81-92 (1996).
7. Y. Oka, S. Koshizuka, “Supercritical-pressure Once-through Cycle Light Water Cooled Reactor Concept,” *Journal of Nuclear Science and Technology*, **38**, pp.1081-1089 (2001).
8. Y. Ishiwatari, Y. Oka, S. Koshizuka, “Control of a High Temperature Supercritical Pressure Light Water cooled and Moderated Reactor with water Rods,” *Journal of Nuclear Science and Technology*, **40**, pp.298-306 (2003).
9. A. Yamaji, Y. Oka, S. Koshizuka, “Three-dimensional Core Design of High Temperature Supercritical-Pressure Light Water Reactor with Neutronic and Thermal-Hydraulic Coupling,” *Journal of Nuclear Science and Technology*, **42**, pp.8-19 (2005).
10. D. Squarer et al., “High performance light water reactor,” *Nuclear Engineering and Design*, **221**, pp.167-180 (2003).
11. J. Buongiorno, P.E. MacDonald, “Supercritical Water reactor (SCWR) – Progress report for the FY-03 Generation-IV R&D Activities for the Development of the SCWR in the U.S.,” INEEL/EXT-03-01210, INEEL, (2003).
12. X. Cheng, T. Schulenberg, D. Bitterman, P. Rau, “Design analysis of core assemblies for supercritical pressure conditions,” *Nuclear Engineering and Design*, **223**, pp.279-294 (2003).
13. T.T. Yi, S. Koshizuka, Y. Oka, “A Linear Stability Analysis of Supercritical Water Reactors, (I) Thermal-Hydraulic Stability,” *Journal of Nuclear Science and Technology*, **41**, pp.1166-1175 (2004).

14. T.T. Yi, S. Koshizuka, Y. Oka, “A Linear Stability Analysis of Supercritical Water Reactors, (II) Coupled Neutronic Thermal-Hydraulic Stability,” *Journal of Nuclear Science and Technology*, **41**, pp.1176-1186 (2004).
15. J.S. Muransky, J.G. Shatford, C.E. Peterson, G.B. Swindlehurst, “Oconee Nuclear Power Station Main Steam Line Break Analysis for Steam Generator Tube Stress evaluation,” *Nuclear Technology*, **148**, pp.48-55 (2004).
16. X. Cheng, T. Schulenberg, “Heat Transfer at Supercritical Pressures – Literature Review and Application to an HPLWR,” FZKA 6609, Forschungszentrum Karlsruhe GmbH, (2001).
17. G. F. Hewitt, *Heat Exchanger Design Handbook 1998*, Begell House Inc., New York, USA (1998).
18. R.A. Matzie, A. Worrall, “The AP1000 reactor – the nuclear renaissance option,” *Nuclear Energy*, **43**, pp.33-45 (2004).
19. K.N. Ivanov et al., “Pressurized Water Reactor Main Steam Line Break (MSLB) Benchmark – Volume 1: Final Specifications,” NEA/NSC/DOC(99)8, OECD Nuclear Energy Agency, (1999).
20. V.H. Sánchez-Espinoza, W. Hering, “Investigations of the Appropriateness of RELAP5/MOD3 for the Safety Evaluation of an Innovative Reactor Operating at Thermodynamically Supercritical Conditions,” FZKA 6749, Forschungszentrum Karlsruhe GmbH, (2003).